NON-PUBLIC?: N
ACCESSION #: 8906140112
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Pilgrim Nuclear Power Station PAGE: 1 OF 09

DOCKET NUMBER: 05000293

TITLE: Automatic Turbine Trip, Generator Trip, and Reactor Scram due to High Reactor Vessel Water Level
EVENT DATE: 05/03/89 LER #: 89-015-00 REPORT DATE: 06/02/89

OPERATING MODE: N POWER LEVEL: 024

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Douglas W. Ellis-Senior Compliance Engineer TELEPHONE: (508)747-8160

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SB COMPONENT: CL MANUFACTURER: A613
REPORTABLE TO NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On May 3, 1989 at 0326 hours, a high Reactor Vessel (RV) water level occurred that resulted in an automatic Turbine trip, Generator trip, and reactor scram at 24 percent reactor power. The event included a designed automatic transfer of the power source for the 4160 VAC Auxiliary Power Distribution System (APDS). During the event one inboard Main Steam isolation valve (MSIV) and four Sampling System isolation valves closed automatically. The high RV water level was primarily caused while troubleshooting the actuator controls of a Feedwater System regulating valve. The Sampling System isolation valves closed because of a transient voltage decrease (APDS transfer) that de-energized the 120 VAC coils of related relays. A random failure of the DC pilot solenoid coil (125 VDC) for the MSIV, together with the effects of the APDS transfer to the MSIV's AC pilot solenoid control relay, caused the MSIV to close.

Significant corrective actions taken include restricted use of the troubleshooting procedure, replacement of some actuator components for both regulating valves, replacement of the failed DC pilot solenoid assembly, and

adjustment of the coil drop out voltage (or coil replacement) for appropriate control relays. The DC pilot solenoid assembly (model 6910-020) was examined by the manufacturer (Automatic Valve Corporation).

This event occurred with the reactor mode selector switch in the RUN position with the Feedwater System in single element (RV water level) control. This report is submitted per 10 CFR 50.73(a)(2)(iv) and this event posed no threat to the public health and safety.

END OF ABSTRACT

EVENT DESCRIPTION

On May 3, 1989 at approximately 0326 hours, a high Reactor Vessel (RV) water level occurred at approximately 24 percent reactor power. The water level resulted in an automatic sequence of designed responses that included a Turbine trip, Generator trip, and a Reactor Protection System (RPS) load reject scram.

The Turbine Trip included the following responses:

Automatic closing of the Main Steam System/Turbine valves (stop valves, control valves, combined intermediate valves) and automatic opening of the Turbine Bypass Valves.

Automatic opening of the Generator Field Breaker.

Automatic opening of the 345 KV air circuit breakers ACB-104 (352-4) and ACB-105 (352-5) in the switchyard.

Automatic transfer of the source of 4160 VAC power for the Auxiliary Power Distribution System (APDS) from the Unit Auxiliary Transformer (UAT) to the Startup Transformer (SUT).

The Generator trip was the designed response to the loss of field that resulted from the automatic opening of the field breaker. The RPS scram was the designed response to the closing (i.e. less than 90 percent open) of the Turbine stop valves.

The RV water level decreased in response to the scram because of shrink, i.e. decrease in the void fraction in the RV water. The RV water level decreased to approximately plus 17 inches.

Initial Control Room licensed operator response to the event was to verify the

Turbine trip and the full insertion of the control rods. The inboard Primary Containment System (PCS)/Main Steam line 'A' isolation valve (MSIV) was observed to have closed automatically during the event. After verifying that only one MSIV had closed, the reactor mode selector switch (RMSS) was moved to the REFUEL position in accordance with procedure 2.1.6, "Reactor Scram". The one feedwater pump in service was secured after the RV water level stabilized. The positions of the Primary Containment Isolation Control System (PCIS) control switches were left as-is to facilitate reconstruction of the event.

Prior to the event, the Feedwater System Train 'B' regulating valve (FV-642B) was to be put into service. The system was being automatically controlled by the master feedwater controller that was in the single element (RV water level) control mode. The Train 'A' regulating valve (FV-642A) was in service in the automatic control mode.

When the valve (FV-642B) was being put into service, Train 'B' flow fluctuations were identified at the Reactor Control Panel C-905. Initial investigation revealed that the flow fluctuations were due to the air operated Train 'B' regulating valve or its controls. The valve (FV-642B) was then removed from service on May 2, 1989 at 2208 hours by adjusting its Panel C-905 controller to the CLOSE position in the manual control mode. The valve's handwheel (manual isolation) was tightened to prevent valve (FV-642B) operation until further investigation was performed by the Instrumentation and Control (I&C) Division. Prior to the investigation, a briefing was conducted by the shift Nuclear Watch Engineer (NWE) with appropriate shift operators and I&C technicians on May 3, 1989 at 0300 hours. The investigation was performed per Procedure 3.M.3-8, "Inspection/Troubleshooting-Electrical Circuits". For the investigation, a meter was connected in series with the valve's (FV-642B) electric-to-pneumatic converter (E/P). The connections were made to monitor the performance of the converter with the valve (FV-642B) in limited operation. The connection resulted in a Panel C-905 alarm, "Feedwater Valve Control Signal Failure". The NWE, stationed near the valve, notified the Control Room operators that the troubleshooting was to begin. Two opening rotations of the valve's handwheel were made. Reportedly, the valve stem suddenly moved open one-half inch at the end of the second handwheel rotation. In contrast, process computer (EPIC) feedwater flow data indicates that a gradual but relatively large increase in Train 'B' flow and a gradual but smaller decrease in Train 'A' flow occurred as a result of the handwheel rotations. A stem travel of approximately 2 (two) inches corresponds to a full valve stroke. The licensed operator at Panel C-905, monitoring the feedwater flow and RV water level, notified the personnel stationed at the valve to reverse their actions. The Panel C-905 operator attempted to compensate for the increasing RV water level by increasing the amount of reject flow from the Reactor Water Cleanup System to the Main Condenser. However, the relatively high opening force exerted by the valve/handwheel spring prevented the operators from manually closing the valve (FV-642B). The high RV water level setpoint (plus 45 inches) was subsequently reached that resulted in the event. The maximum RV water level that occurred was approximately plus 48 inches.

A post trip review was conducted per Procedure 1.3.37 (Rev. 4), "Post Trip Reviews". During the review four PCS Group two/Sampling System isolation valves (SV-5065-11A, -14A, -20B, and -70) were identified to have closed during the event in addition to the inboard PCS Group one/steam line 'A' MSIV (AO-203-IA). The PCS isolation valves are automatically controlled by the PCIS. Because the reason for the closing of the Sampling System isolation valves and the MSIV could not be identified at that time, multi-disciplinary teams were formed to investigate.

Failure and Malfunction Report (F&MR) 89-181 was written to document the problem with the Feedwater System Train 'B' regulating valve. F&MR 89-182 was written to document the event and closing of the MSIV. The NRC Operations Center was notified on May 3, 1989 at 0430 hours. F&MR 89-183 was written to document the closing of the four Sampling System isolation valves. The NRC Operations Center was notified on May 3, 1989 at 0620 hours.

This event occurred at approximately 24 percent reactor power with the RMSS in the RUN position. The Reactor Vessel (RV) pressure was approximately 940 psig with the RV water temperature at 537 degrees Fahrenheit. The automatic transfer selector switches of the 4160 VAC switchgear were in the ON position. Both 345 KV transmission lines were in service.

CAUSE

For clarity, the cause(s) are described as follows:

High Reactor Vessel (RV) Water Level

The primary cause for the high RV water level was the use of the general troubleshooting procedure (3.M.3-8) for troubleshooting the Feedwater System Train 'B' regulating valve (FV-642B) actuator controls. Because the procedure did not require a detailed work plan, the valve's air lockout was not reset per procedure 3.M.2-10 (Feedwater Control Valve Isolation and Maintenance) prior to moving the handwheel. Contributing factors included a mis-oriented diaph
agm (actuator dome) pressure gauge, and an air leak in the valve actuator's air lockout device and actuator diaphragm. Prior to moving the

handwheel, the actuator's diaphragm pressure was incorrectly read by an I&C Technician as normal (approximately 100 psig) when the pressure was approximately zero psig. The pressure was incorrectly read because the diaphragm pressure gauge was oriented oppositely from the other actuator pressure gauges (positioner output, E/P output, and air supply). Because the pressure was incorrectly read, the actuator pressure was believed to be correct. During investigation after the event, the actuator was pressurized and monitored for stability. The pressure decreased from normal operating pressure (approximately 100 psig) to zero psig in approximately 12 minutes. The pressure decrease was due to the combined air leakage from the valve actuator's air lockout device and actuator diaphragm.

Automatic Closing of the Sampling System Isolation Valves

The four Sampling System isolation valves closed because of a transient voltage decrease that de-energized the 120 VAC coil(s) of related PCIS tripping and/or control relays. The voltage decrease was the result of the APDS power source transfer. The automatic closing of the isolation valves was not the result of an accident mitigating signal(s).

The transfer function of the APDS safety-related 4160 VAC Busses A-5 and A-6 was tested. The testing was conducted to determine if a slow transfer time could have resulted in a voltage decrease of sufficient magnitude and duration to cause an actuation(s) of a PCIS relay(s). A dead Bus A-5 transfer test was conducted per procedure TP 89-46, "Timing for Unit Aux Fast Transfer". The test results revealed that the time from the transfer initiating signal to the closing of the SUT/Bus A-5 breaker was within the expected range of 7 cycles (Hertz) to 9 cycles, i.e. 117 to 150 milli-seconds. The results provided high confidence that the UAT/Bus A-5 (and similar Bus A-6) breaker and SUT/Bus A-5 (and similar Bus A-6) breaker operated as designed during the event. Live transfer tests of Busses A-5 and A-6 were conducted per Procedure TP 89-48, "Functional Test of 4 KV Fast Transfer". High speed recorders were used to monitor safety-related Busses (4160 VAC, 480 VAC, and 120 VAC) for transfer time and resultant voltage levels. The tests revealed that the UAT/Bus A-5 to SUT/Bus A-5 transfer and the UAT/Bus A-6 to SUT/Bus A-6 transfer was five to five and one-half cycles. The transfer was timed from the opening of the UAT/Bus A-5 (and Bus A-6) breaker(s) to the closing of the SUT/Bus A-5 (and Bus A-6) breaker(s). The live transfers were within the expected transfer time for the Busses. The tests resulted in a transient voltage decrease to slightly less than 65 percent of the normal voltage on some of the Busses. PCIS relays found in the tripped condition after the May 3, 1989 event were tripped as a result of the tests. The Control Room alarm responses were correct and consistent with the tests.

In-situ testing of selected PCIS relays was conducted per procedure TP 89-49, "Relay Test of 16A-K17X4, 16A-K18X2, 16A-K18X5, 16A-K17, 16A-K17X, and 16A-K26". The testing included measurement of the coil dropout voltage(s) for the relays. The test results, in conjunction with the results of the live 4160 VAC Bus A-5 and Bus A-6 testing, confirmed that the partial actuations of the PCIS relays were caused by a (momentary) transient voltage decrease (due to the APDS power source transfer). The voltage decrease was sufficiently close to the coil dropout voltage of the relays to de-energize the 120 VAC coil(s) of PCIS tripping and/or control relays that are associated with the four Sampling System isolation valves.

The Analog Trip System (ATS) response to the event was analyzed by the Nuclear Engineering Department as a possible cause for the partial PCIS actuations. The analysis concluded that the partial actuations were not caused by the ATS transmitters or ATS trip units. The analysis included the use of data from the process computer (EPIC), alarm typer printouts and review of ATS setpoint calibrations. The review revealed highly stable setpoints for the transmitters and trip units.

Automatic Closing of the MSIV (AO-203-1A)

The MSIV closed because of the apparent previous failure of the valve's DC pilot solenoid coil together with a transient voltage decrease (APDS transfer) that de-energized the 120 VAC coil of the PCIS control relay (16A-K14) associated with the valve's AC pilot solenoid. Each MSIV is maintained in the open position by pneumatic pressure that is controlled by the valve's pilot solenoids. Each MSIV is designed to close automatically if both of the valve's pilot solenoids (one AC and one DC) become de-energized. The APDS power source transfer resulted in a transient voltage decrease that caused the normally energized 120 VAC coils of the inboard PCIS control relay 16A-K14 and outboard PCIS control relay 16A-K16 to become de-energized. The 16A-K14 and 16A-K16 relays (General Electric type HFA) function to control the AC pilot solenoids for the inboard and outboard MSIVs, respectively. All of the inboard and outboard AC pilot solenoids became de-energized because the control relays became de-energized (due to the APDS transfer). The inboard steam line 'A' MSIV closed because the coil of its DC pilot solenoid had failed (apparently prior to the event). As-found, the DC pilot solenoid coil was electrically open prior to removal. The free movement of the solenoid plunger indicated that the coil failure was not caused by plunger binding. The coil exhibited no evidence of overheating. The solenoid coil is the same type installed since 1978 without previous failure. The coil's maximum supply voltage (132 to 134VDC) is within the coil's specified operating range (106.25 to 137.5 VDC). The solenoid assembly (model 6910-020) was sent to the manufacturer (Automatic Valve Corporation) for examination of the coil.

Disassembly of the coil revealed no separation of the lead wire from the coil. Therefore, the cause for the DC pilot solenoid coil failure is attributed to random failure. The DC pilot solenoid data is as follows: 125 VDC, (ORIF) 3/32, (ID) PAR, 11 watts.

CORRECTIVE ACTION

Corrective actions taken or planned include:

Disassembly, inspection, and replacement of components for the actuator of the Feedwater System Train 'A' regulating valve (FV-642A) and Train 'B' regulating valve (FV-642B).

Recalibration of selected Feedwater System components and controls.

Replacement of the failed DC pilot solenoid assembly for the inboard steam line 'A' MSIV (AO-203-1A).

Inspection and testing of the inboard and outboard DC pilot solenoids.

Insulation testing and walkdown of the cabling for the inboard and outboard MSIVs. The cables for the outboard MSIV pilot solenoids (AC and DC) were replaced. The cables were replaced because the insulation for some of the cables was degraded. Though degraded, the cables satisfactorily passed insulation testing (meggaring).

Adjustment of the coil dropout voltage for the inboard and outboard PCIS relays 16A-K14 (Panel C-941) and 16A-K16 (Panel C-942) to approximately 60 VAC. The coil dropout voltage adjustment reduces the likelihood of an actuation(s) of the relays (and related MSIV AC pilot solenoids) if a similar APDS power source transfer occurs in the future. The MSIV DC pilot solenoids are controlled by relays having direct current coils.

Replacement of the 120 VAC coil with a 115 VAC coil for the inboard and outboard PCIS relays 16A-K17 (Panel C-941) and 16A-K18 (Panel C-942). The coil replacements decrease the dropout voltage by approximately 5 (five) volts, and reduces the likelihood of an unnecessary actuation(s) of the relays if a similar APDS power source transfer occurs in the future. The dropout voltage for the 16A-K17 and 16A-K18 relays (General Electric type CR120A) is not adjustable.

Revision of the reactor scram response procedure (2.1.6). The revision included licensed operator guidance if a loss of power to the 120 VAC portion of the logic circuitry for the MSIVs occurs.

Planned revision of Procedure 3.M.3-8 (currently Rev. 13) to restrict the use of the procedure. Interim measures taken to restrict the use of the procedure includes approval by the Plant Department Manager (or designee) prior to use for safety-related work or the Maintenance Section Manager for nonsafety-related work.

Planned addition of the Feedwater System regulating valves and controls into the routine preventive maintenance program.

SAFETY CONSEQUENCES

This event posed no threat to the public health and safety.

The automatic Turbine and Generator trips were appropriate designed responses to the high RV water level. The automatic opening of the 345 KV air circuit breakers was the designed response to the Generator trip. The automatic transfer of the power source for the APDS was the designed response to the Turbine trip. The RPS scram was the designed response to the closing (i.e. less than 90 percent open) of the Turbine stop valves.

The APDS power source transfer resulted in a transient voltage decrease on safety-related busses (4160 VAC, 480 VAC, and 120 VAC). The voltage response due to the transfer has been previously evaluated in the Final Safety Analysis Report and is bounded by a total loss of offsite power event.

The APDS transfer time was approximately 150 milli-seconds or less. The transfer time was much less than the one second time period that the RPS motor-generator sets can independently maintain sufficient 120 VAC power to prevent an RPS initiated reactor scram due to an APDS transfer.

This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv) because the RPS logic circuitry was actuated.

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station Licensee Event Reports (LERs) submitted since January 1984. The review was focused to LERs submitted in accordance with 10 CFR 50.73(a)(2)(iv) that involved a high RV water level. The review identified high RV water level events reported in LERs 50-293/84-020-00, 85-014-00, 88-024-00, and 89-007-00.

For LER 84-020-00, a PCIS Group 1 (one) isolation signal occurred due to a high RV water level. The isolation signal resulted in the automatic closing of the MSIVs. At the time of the event, the reactor power level was approximately one percent, the RMSS was in the STARTUP position, the RV pressure was approximately 100 psig, the RV water level was being manually controlled, and the Residual Heat Removal System (RHRS) was in the Suppression Pool Cooling (SPC) mode. The cause for the high water level was attributed to leakage of RHRS/SPC water past the seat of an RHRS valve and into the RV.

For LER 85-014-00, a PCIS Group 1 (one) isolation signal and a full RPS scram signal occurred due to a high RV water level. The isolation signal resulted in the automatic closing of the MSIVs. At the time of the event, the reactor power level was approximately 10 percent, the RMSS was in the STARTUP position, the RV pressure was approximately 700 psig, and the RV water level was being manually controlled. The cause for the high water level was attributed to utility licensed operator error. The scram signal (and reactor scram) occurred when the RV pressure was approximately 700 psig. A high water level in the RV (i.e. closing of the MSIVs) results in a scram signal when the RMSS is in the REFUEL or SHUTDOWN or STARTUP position, and if the RV pressure is greater than approximately 600 psig.

For LER 88-024-00, a PCIS Group 1 (one) isolation signal occurred due to a high RV water level. The isolation signal resulted in the automatic closing of the MSIVs. At the time of the event, the reactor power level was zero percent, the RMSS was in the REFUEL position for a surveillance activity, the RV pressure was zero psig, and the RV water level was being manually controlled. The cause for the high water level was attributed to a pin that became disassociated from the feedback cam linkage of the positioner for the Feedwater System Train 'A' regulating valve (FV-642A).

For LER 89-007-00, a PCIS Group 1 (one) isolation signal occurred due to a high RV water level. The isolation signal resulted in the automatic closing of the inboard MSIVs. The outboard steam line 'C' MSIV closed automatically and the other outboard MSIVs remained closed. At the time of the event the reactor power level was approximately 0.8 percent, the RMSS in the STARTUP position, and the RV pressure was approximately 278 psig. The MSIVs were being tested with a differential pressure of approximately 150 psid across the seat of the MSIV(s). The high RV water level was caused by the swell (expansion) of RV water that occurred when the outboard line 'C' MSIV was opened for testing. The cause for the event was attributed to an inadequacy in the development and review of the approved test procedure and a relatively fast opening time for the MSIV.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

COMPONENTS CODES

Bus BU
Cable, Low-voltage Power CBL4
Circuit Breaker, AC 52
Circuit Breaker, Field 41
Coil CL
Control Operator, Flow FCO
Valve, Isolation ISV
Valve, Control, Flow (FV-642B) FCV
Generator, Turbine TG
Relay RLY
Relay, Tripping 94
Solenoid SOL
Switchgear SWGR

SYSTEMS

Containment Isolation Control System (PCIS) JM
Engineered Safety Features Actuation System JE
(PCIS/RPS)
Feedwater System SJ
Integrated Control System JA
Main Generator System TB
Main Steam System SB
Main Turbine System TA
Post Accident Monitoring System IP
Switchyard System FK
Turbine Supervisory Control System JJ
Main Generator Output Power System (345KV) EL
Medium-Voltage Power System (4160 VAC) EA
Plant Protection System (RPS) JC

ATTACHMENT 1 TO 8906140112 PAGE 1 OF 1

10 CFR 50.73

BOSTON EDISON
Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, Massachusetts 02360

Ralph G. Bird
Senior Vice President - Nuclear

June 2, 1989
BECo Ltr. 89-075

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Docket No. 50-293
License No. DPR-35

Dear Sir:

The attached Licensee Event Report (LER) 89-015-00, "Automatic Turbine Trip, Generator Trip, and Reactor Scram due to High Reactor Vessel Water Level", is submitted in accordance with 10 CFR Part 50.73.

Please do not hesitate to contact me if there are any questions regarding this report.

R. G. Bird

DWE/bal

Enclosure: LER 89-015-00

cc: Mr. William Russell
Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Rd.
King of Prussia, PA 19406

Sr. NRC Resident Inspector - Pilgrim Station

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